NON-PUBLIC?: N
ACCESSION #: 9111070117
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Calvert Cliffs, Unit 1 PAGE: 1 OF 04

DOCKET NUMBER: 05000317

TITLE: Reactor Protection System Actuation and Plant Trip Due to Low Steam Generator Water Levels Caused by Loose Electrical Fuse
EVENT DATE: 10/01/91 LER #: 91-003-00 REPORT DATE: 10/31/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 093

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Edward P. Wilson, Compliance Engineer TELEPHONE: (301) 260-2062

COMPONENT FAILURE DESCRIPTION:
CAUSE: A SYSTEM: JK COMPONENT: FUB MANUFACTURER: M080
REPORTABLE NPRDS: YES

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On October 1, 1991 Calvert Cliffs Unit 1 tripped from 93 percent power due to low Steam Generator (SG) water levels. The low SG water levels resulted from a loose fuse in the power supply to 12 Feedwater Regulating Valve (FRV) differential pressure (dp) controller and 12 Steam Generator Feed Pump (SGFP) turbine speed controller. A bent fuse clip caused an electrical discontinuity. The fuse holder has been replaced and personnel whose duties involve fuse manipulation will be trained on proper fuse installation and removal.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On October 1, 1991 Calvert Cliffs Unit 1 tripped from 93 percent power due to low Steam Generator (SG) water levels. The low SG water levels resulted from a loose fuse in the power supply to 12 Feedwater Regulating Valve (FRV) differential pressure (dp) controller and 12 Steam Generator Feed Pump (SGFP) turbine speed controller. A bent fuse clip caused an electrical discontinuity. The fuse holder has been replaced and personnel whose duties involve fuse manipulation will be trained on proper fuse installation and removal.

During standard operation, an auctioneered FRV dp signal from either 11 or 12 FRV dp controller is used by the SGFP turbine speed controllers to maintain a constant dp across the FRVS. In the normal configuration, 12 FRV dp controller is in automatic and 11 FRV dp controller is manually set to provide a lower 25 percent signal. This is done to prevent system oscillation and promote stable system response.

On October 1, 1991 at approximately 1047 hours, while operating at 93 percent power, electrical power was lost to 1-PDIC-4517 (12 FRV dp controller) and 1-HIC-4517 (SGFP 12 turbine speed controller). Upon loss of power to the speed controller for 12 Steam Generator Feed Pump (SGFP), its speed and flow output began to ramp down rapidly. In addition, the loss of 12 FRV dp controller resulted in the input of a lower 25 percent signal to 11 SGFP speed controller. The combined effect was a sharp reduction in feedwater flow. With the plant operating at 93 percent power, Steam Generator (SG) water levels fell rapidly.

The Control Room Operator observed the low level alarms for both 11 and 12 SGs and observed levels decreasing rapidly in both. He also noticed the loss of power to 1-PDIC-4517 and 1-HIC-4517. Unable to control 12 SGFP speed, he attempted to take manual control of 11 SGFP but was unable to do so before the SG low level setpoint was reached and the reactor tripped at 1048. The appropriate Emergency Operating Procedures were performed without incident. Auxiliary Feedwater flow was initiated to restore and maintain SG level in accordance with specified core and Reactor Coolant System heat removal alternate action.

Immediately following the trip, technicians used electrical schematics to ascertain a common power supply for the deenergized components and, using approved troubleshooting techniques, began checking the electrical supply path. As part of this effort, technicians attempted to take voltage readings across fuse 1-1C03-UZZZ. When a technician touched the top of the fuse with the voltmeter probe, sparks were observed and power was restored at that instant to the previously deenergized components. Subsequent removal of the fuseholder revealed that the upper clip was slightly deformed, permitting a small gap to exist between the fuse clip

and the upper end of the fuse. This gap was sufficient to cause electrical discontinuity in the electrical supply path to the two controllers.

II. CAUSE OF EVENT

The root cause of this event was a bent upper fuse clip in the fuseholder which held fuse 1-1C03-UZZZ. The electrical discontinuity that resulted from this caused a loss of 12 SGFP speed control and a reduction in 11 SGFP output which in turn caused a reactor trip due to low SG water levels. Given the extremely small clearances between the fuse clip and the upper end of the fuse; dust, dirt, or vibration could have caused the circuit to be broken.

Preliminary material analysis of the fuse clip indicated that it had lost its stiffness and thus would not return to its proper shape when flexed. It appears this loss of stiffness was due to improper fuse installation or removal at some time in the past. A survey conducted following this event found a small percentage of fuses installed improperly in that the fuse clips were spread too far. In addition, access to the fuse involved in this event is very difficult and involves very little room in which to utilize fuse pullers. As a result, extra care is required to ensure proper installation and removal.

III. ANALYSIS OF EVENT

The Calvert Cliffs Updated Final Safety Analysis Report describes the simultaneous full closure of both Feedwater Regulating Valves (FRVs) from 102 percent power as its worst case loss of feedwater accident. Using these initial conditions, the analysis concludes that no limits are exceeded. Considering that, this event was initiated from 93 percent power, neither FRV went shut until after the trip, some flow from 11 SGFP was available, and the protective system operated as expected, there was no safety significance associated with this event.

There were no other components or systems which were inoperable and/or out of service which contributed to this event. No plant systems or other component failures resulted from this event.

The event is reportable in accordance with 10 CFR 50.73(a)(2)(iv), "any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (EFS), including the Reactor Protection System (RPS)."

IV. CORRECTIVE ACTIONS

Plant Operators took immediate corrective action following the trip in accordance with Emergency Operating Procedures EOP-0 (Post-Trip Immediate Actions) and EOP-1 (Reactor Trip) to stabilize the plant.

The fuse holder and fuse 1-2C03-UZZZ were replaced. A survey of other control room fuses was conducted and the small percentage which were found improperly installed were corrected.

To prevent recurrence, initial training and/or qualification for plant operator electricians, instrumentation technicians, and safety taggers will be amended t include knowledge of proper fuse installation and removal. This event will also be reviewed as part of continuing training for these personnel.

V. ADDITIONAL INFORMATION

A. Identification of Components Referred to in this LER:

IEEE 803 IEEE 805
EIIS Funct System ID

Steam Generator SG N/A
Steam Generator Feed Pump P SJ
Feedwater Regulating Valve LCV SJ
Speed Controller SC JK
Differential Pressure Controller PC JK
Fuse FU JK
Fuse Holder FUB JK
Reactor Coolant System N/A AB
Reactor Protection System N/A JC

B. Previous Similar Events

There have been no previous similar events.

BALTIMORE
GAS AND
ELECTRIC

CHARLES CENTER o P.O. BOX 1475 o BALTIMORE, MARYLAND 21203-1475

R. E. DENTON
GENERAL MANAGER
CALVERT CLIFFS
October 31, 1991

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1; Docket No. 50-317; License No. DPR 53
Licensee Event Report 91-003

Gentlemen:

The attached report is being sent to you as required under 10 CFR 50.73 guidelines. Should you have any questions regarding this report, we will be pleased to discuss them with you.

Very truly yours,

RED/EPW/bjd
Attachment

cc: D. A. Brune, Esquire
J. E. Silberg, Esquire
R. A. Capra, NRC
D. G. McDonald, Jr., NRC
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A. G. Howe, NRC
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